



A PECO Energy/British Energy Company

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Docket No. 50-461

10CFR50.73

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station
Licensee Event Report No. 2000-001-00

Dear Madam or Sir:

Enclosed is Licensee Event Report (LER) No. 2000-001-00: Operation of Mislabeled Switch While Performing Preventive Maintenance on a Circuit Breaker Results in Loss of 4160 Volt 1B Bus, Reactor Water Level Transient and Manual Scram. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

Michael T. Coyle
Vice President

JRF/blf

Enclosure

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-960
NRC Region III, Regional Administrator

RGH-001

JE22

LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

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FACILITY NAME (1)

Clinton Power Station

DOCKET NUMBER (2)

05000461

PAGE (3)

1 OF 5

TITLE (4)

Operation of Mislabeled Switch While Performing Preventive Maintenance on a Circuit Breaker Results in Loss of 4160 Volt 1B Bus, Reactor Water Level Transient and Manual Scram

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
05	17	2000	2000	001	00				None	05000
									FACILITY NAME	DOCKET NUMBER
									None	05000
									FACILITY NAME	DOCKET NUMBER
									None	05000
OPERATING			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)							
1			20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(i)	
			20.2203(a)(1)			20.2203(a)(3)(i)			50.73(a)(2)(ii)	
POWER LEVEL (10)			20.2203(a)(2)(i)			20.2203(a)(3)(ii)			50.73(a)(2)(iii)	
100			20.2203(a)(2)(ii)			20.2203(a)(4)			X 50.73(a)(2)(iv)	
			20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)	
			20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)	
									OTHER	
									Specify in Abstract below or in NRC Form 366A	

LICENSEE CONTACT FOR THIS LER (12)

NAME

J. R. Hays, Operations Support Manager

TELEPHONE NUMBER (Include Area Code)

(217) 935-8881, Extension 3692

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE).		X	NO	EXPECTED	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 17, 2000, the plant was in Mode 1, at approximately 100 percent power. Functional testing of over current relays for the reserve feeder breaker to the 4160 Volt (V) 1B Bus (non-safety) was in progress. While inputting a simulated over current condition to test the over current relays on the reserve feeder breaker, the main feeder breaker to the 4160 V 1B Bus tripped unexpectedly resulting in a loss of power to the 4160 V 1B Bus. The resulting loss of power caused a loss of feedwater flow and reactor vessel level transient. In response to the level transient, a manual reactor scram was initiated and the reactor shut down. The cause of this event was incorrect labeling of a test switch on the 4160 V 1B Bus. The corrective actions for this event include performing walkdowns of other electrical busses to identify labeling problems, briefing maintenance personnel on this event, and incorporating a Labeling Validation Form in Electrical Maintenance work packages.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Clinton Power Station	05000461	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 5
		2000	- 001	- 00	

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

DESCRIPTION OF EVENT

On May 17, 2000, the plant was in Mode 1, at approximately 100 percent power. Auxiliary Boilers [VK] "A" and "B" were secured for maintenance. Functional testing was in progress on the reserve feeder breaker [BKR] to the 4160 volt (V) (non-safety) 1B Bus [BU]. At approximately 1014 hours, Main Control Room (MCR) personnel identified a loss of the 4160 V 1B Bus and a lowering reactor [RCT] vessel water level. In response to the lowering reactor vessel water level, the Reactor Operator initiated a manual reactor scram by placing the mode select switch in shutdown.

Functional testing of the over current relay [RLY] protection for the reserve feeder breaker to the 4160 V 1B Bus was being performed to satisfy a portion of preventive maintenance task PEMAPM558. This testing was being performed in accordance with CPS procedure 8501.88, "4.16kV Bus 1B Feed Breaker Protection Relays Functional Testing." (This was the first time that this procedure had been used for functional testing the reserve feeder breaker over current relay protection.) On May 17, 2000, at approximately 0823 hours, the power source to the 4160 V 1B Bus was transferred from the reserve to the main power source in preparation for the test. The reserve feeder breaker was then racked out to the test position without incident. To prevent actuation of the main feeder breaker to the 4160 V 1B Bus during this test, CPS procedure 8501.88 requires opening blades "A" and "H" on test switch 2TS-221B. This operation interrupts over current trip circuitry to the main feed breaker [BKR] during the functional test. The technicians performing the maintenance opened blades "A" and "H" on the test switch labeled 2TS-221B in accordance with the procedure; however, the test switch was mislabeled and blades "A" and "H" were opened on the wrong test switch (2TS-221B-1). At approximately 1014 hours, the technicians then manually initiated a simulated over current condition on the "A" phase overcurrent relay. At this time the main feeder breaker to the 4160 V 1B Bus unexpectedly tripped resulting in a loss of power to the 4160 V 1B Bus.

The loss of power to the 4160 V 1B Bus resulted in the deenergization of the following major components; 2 of the 3 operating Condensate Booster Pumps, 1 of the 3 operating Condensate Pumps, the operating Service Air Compressor, 1 of the 2 operating Component Cooling Water Pumps, the operating Service Water Pump, and the operating Control Rod Drive Pump. The loss of the Condensate Booster Pumps and the Condensate Pump caused the operating Reactor Feed Pumps to trip on low suction pressure. MCR operators immediately recognized the loss of the equipment and identified a lowering reactor water level as a result of the loss of feedwater. At approximately 1014 hours, in response to the plant transient, the Control Room Supervisor (CRS) directed the Reactor Operator (RO) to manually scram the reactor and the RO initiated a manual reactor scram by placing the mode select switch in the shutdown position. Upon initiation of the scram, all control rods fully inserted and the reactor shut down.

Immediately following the reactor scram, reactor vessel level and pressure were maintained by venting steam to the main condenser through the main steam bypass valves and injecting with feedwater/condensate.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Clinton Power Station	05000461	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	3 OF 5
		2000	- 001	- 00	

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Efforts to perform a reactor cool down to allow entry into Mode 4 were complicated by a loss of service/instrument air [LF] [LD]. Due to the loss of power to the operating service air compressor, service/instrument air system pressure began to decrease.

The standby service air compressor failed to auto start due to low component cooling water (CCW) [CC] pressure caused by the loss of power to one of the two operating CCW Pumps. As a result of the loss of service/instrument air, the Main Steam Isolation Valves (MSIVs) [ISV] started to drift closed (MSIVs are air-assist open, spring close type valves) and at approximately 1028 hours, the outboard MSIVs went fully closed. In order to maximize the amount of steam sent to the main condenser, the main steam line drains and the main steam bypass valves were opened.

The loss of service/instrument air also resulted in a loss of gland sealing steam [TC] to the main turbine. The extraction steam supply valves (air-assist open, spring close type valves) to the steam seal evaporator failed closed due to the loss of service/instrument air. In addition, the backup gland seal steam supply, the Auxiliary Boilers, were tagged out for maintenance. As a result, gland sealing steam to the main turbine was lost. The loss of gland sealing steam to the turbine resulted in the inability to maintain main condenser vacuum and use the main condenser as a heat sink for the reactor. In anticipation of a loss of main condenser vacuum, control room personnel manually closed the Group 1 containment isolation valves [ISV] and broke main condenser vacuum. The loss of condenser vacuum caused a Group 1 isolation signal at 1056 hours.

After the loss of the main condenser as a heat sink, pressure and level control of the reactor were maintained by running Reactor Core Isolation Cooling (RCIC) [BN] and manually cycling Safety Relief Valves.

During the event anomalies were experienced with the operation of the Reactor Water Cleanup (RWCU) system [CE]. Due to the loss of instrument/service air, the isolation valves (air-assist open, spring close type valves) for the RWCU filter demineralizers failed closed. When the valves failed closed, the operating RWCU pumps tripped on low flow as designed. Upon restoration of service/instrument air, operations personnel attempted to return a RWCU pump to operation and several unexpected low flow pump trips occurred. In addition, an invalid Group 4 containment isolation of the RWCU system also occurred. The pump trips and the Group 4 containment isolation of the RWCU system occurred during SRV operation. Once SRV operations was discontinued, further system anomalies were not experienced. Condition report 2-00-05-079 was initiated to investigate the anomalies in operation of the RWCU system.

In addition to the anomalies experienced with the operation of the RWCU system during the event, valve 1E12-F009, Shutdown Cooling Inboard Isolation, failed to open remotely (from the MCR) while attempting to place the Residual Heat Removal (RHR) system [BO] in shutdown cooling. When valve 1E12-F009 failed to open remotely, an equipment operator was dispatched to the valve and the valve was manually opened. Condition Report 2-00-05-083 was initiated to investigate this issue.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Clinton Power Station	05000461	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	4 OF 5
		2000	- 001	- 00	

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Below is a list of other components that did not function as anticipated during this event but did not contribute to the significance of this event.

- Valve 1FW003B, Reactor Feedpump Discharge Bypass, went fully open with a zero percent demand signal. Action request (AR) F18258 was written to investigate and correct this deficiency.
- Valve 1GS041, Auxiliary Steam Supply to Steam Seal Header, did not open electrically. AR F18252 was written to investigate and correct this deficiency.
- Valve 1B33-F067B, Reactor Recirculation Pump 1B Discharge Valve, did not have full closed indication when the valve was closed. AR F18259 was written to investigate and correct this deficiency.
- Heater 1B Flash Tank Level Indication pegged high isolating the 1B Heater and Drain Tank. AR F18255 was written to investigate and correct this deficiency.
- Cycled Condensate (CY) Transfer Pumps "A" and "C" failed to manually start. ARs F11583 and F11582 were written to investigate and correct this deficiency.

No other automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. No other equipment or components were inoperable at the start of the event to the extent that their inoperable condition contributed to this event.

CAUSE OF EVENT

The cause of the event was incorrect labeling of plant equipment. Test switches on the 4160 V 1B Bus were labeled incorrectly. Test switch 2TS-221B was mislabeled as TS-221B and test switch 2TS-221B-1 was mislabeled as 2TS-221B. Due to the incorrect labeling, technicians performing functional testing of the over current protective relays on the reserve feeder breaker to the 4160 V 1B Bus operated the wrong test switch. Operation of the incorrect test switch resulted in deenergization of the 4160 V 1B Bus, a reactor vessel level transient and manual reactor scram.

CORRECTIVE ACTIONS

Test switches 2TS-221B and 2TS-221B-1 were properly labeled.

A walkdown of 4160 V and 6900 V switchgears was performed to identify additional labeling discrepancies. Labeling discrepancies identified during the walkdown were corrected.

A walkdown of safety and non-safety 480 V unit substations will be performed to identify labeling discrepancies. Maintenance personnel will be briefed on this event. A discussion on how labeling discrepancies are to be identified will be incorporated into this briefing.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Clinton Power Station	05000461	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	5 OF 5
		2000	- 001	- 00	

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The Maintenance Planning group has incorporated a "Labeling Validation Form" in Electrical Maintenance work packages. This form is included in packages which implement the first time performance of any new or revised maintenance procedure, including surveillance tests and preventive maintenance items.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv) due to the manual initiation of the reactor protection system [JC]. Assessment of the safety consequences and implication of this event identified that this event was not nuclear safety significant. The event was reviewed for the analyzed transients discussed in Chapter 15 of the Updated Safety Analysis Report (USAR). The analysis determined that this event was within the design basis of the plant.

ADDITIONAL INFORMATION

Clinton Power Station has not reported previous events where incorrect labeling of a component resulted in the misoperation of plant equipment and a plant transient in recent history.

For further information on this event, contact J. R. Hays, Operations Support Manager, (217) 935-8881, extension 3692.